30th NENE 2021
September 6 - 9
BLED-SLOVENIA
International Conference Nuclear Energy for New Europe

Pressurized Thermal Shock Preliminary Analyses of a 2-Loop Pressurized Water Reactor under Loss-Of-Coolant Accident Scenarios

Oriol Costa Garrido, Andrej Prošek, Leon Cizelj
Jožef Stefan Institute
Jamova cesta 39
SI-1000 Ljubljana, Slovenia
oriol.costa@ijs.si, andrej.prosek@ijs.si, leon.cizelj@ijs.si

ABSTRACT

The lifetime of a nuclear power plant strongly depends on the ability of the reactor pressure vessel (RPV) to safely endure reactor transients that may jeopardise its structural integrity. Examples of such transients are loss-of-coolant accidents (LOCA) followed by pressurized thermal shock (PTS) events which are characterized by rapid cooling of the internal RPV surface in combination with repressurization. As the RPV gets embrittled due to neutron irradiation, this ability decreases together with its material's resistance to crack propagation. Therefore, it is of interest to investigate available margins, in terms of embrittlement reference-temperature shifts, under PTS conditions. In this paper, preliminary PTS analyses of a 2-loop pressurized water reactor under selected LOCA scenarios are performed and the methodology to evaluate temperature margins is developed. The thermal-hydraulic and structural analyses are performed, respectively, with RELAP5 and FAVOR computer codes. The results of the preliminary analyses suggest that temperature margins are probably sufficient for the safe and long-term operation of the plant. Nevertheless, the margins obtained for a 12 inch break are rather close to regulatory limits.

1 INTRODUCTION

Pressurized thermal shock (PTS) concerns the possibility that a reactor pressure vessel (RPV) experiences a transient event that subjects its internal surface containing a crack-like defect to a severe thermal shock (or rapid cool down) in combination with pressure [1]. A loss-of-coolant accident (LOCA) due to breaks in primary piping is a type of transient that may lead to PTS due to the injection of cold emergency water required to flood the reactor core. Under these circumstances, a loss of structural integrity of the RPV could occur due to brittle propagation of the crack through the RPV wall. While this type of events is considered in plant safety analyses, it may become life-limiting as the RPV ages [2]. The dominant ageing mechanism with implications in PTS events is embrittlement, which reduces the fracture toughness (or resistance to crack propagation) of the RPV wall material with neutron radiation. Thus, the PTS issue is considered an important cornerstone for the long-term operation of nuclear power plants beyond their design lifetime.

A PTS analysis involves multiple disciplines such as thermal-hydraulics (TH), heat transfer, structural and fracture mechanics, and material science [3,4]. The TH analysis of the reactor transient aims at obtaining the temperature, pressure and heat transfer coefficient in the downcomer region surrounding the core. These are the inputs to the subsequent thermo-mechanical analysis to compute the temperatures and stresses in the RPV wall, which, in the fracture mechanics analyses, serve to evaluate the stress intensity factor (SIF) histories of postulated cracks at the inner surface of the RPV. Finally, the distance between the obtained SIF curve and the fracture toughness curve of the RPV wall material dictates the available shift in ductile-to-brittle transition temperature, i.e. the PTS temperature margin [5].

This paper presents the preliminary PTS analyses of a 2-loop pressurized water reactor (PWR) under loss-of-coolant accidents due to breaks in primary piping. The aim of the work is to develop the analyses methodology (from TH analysis to evaluation of temperature margin) needed to assess the safe and long-term operation of the plant from a PTS standpoint. Following this introduction, the TH analyses are presented in Section 2. In Section 3, the structural integrity analyses are described and the results for the selected accident scenarios are presented in subsections divided into thermo-mechanical, fracture mechanics and temperature margin analyses. At the end, the conclusions are drawn in Section 4.

2 THERMAL-HIDRAULIC ANALYSES

The TH analyses are performed with the latest Reactor Excursion and Leak Analysis Program (RELAP5) code [6] using an available input deck, shown in Fig. 1, of a Westinghouse type 2-loop PWR developed for cladding peak temperature studies during eventual accident scenarios [7].

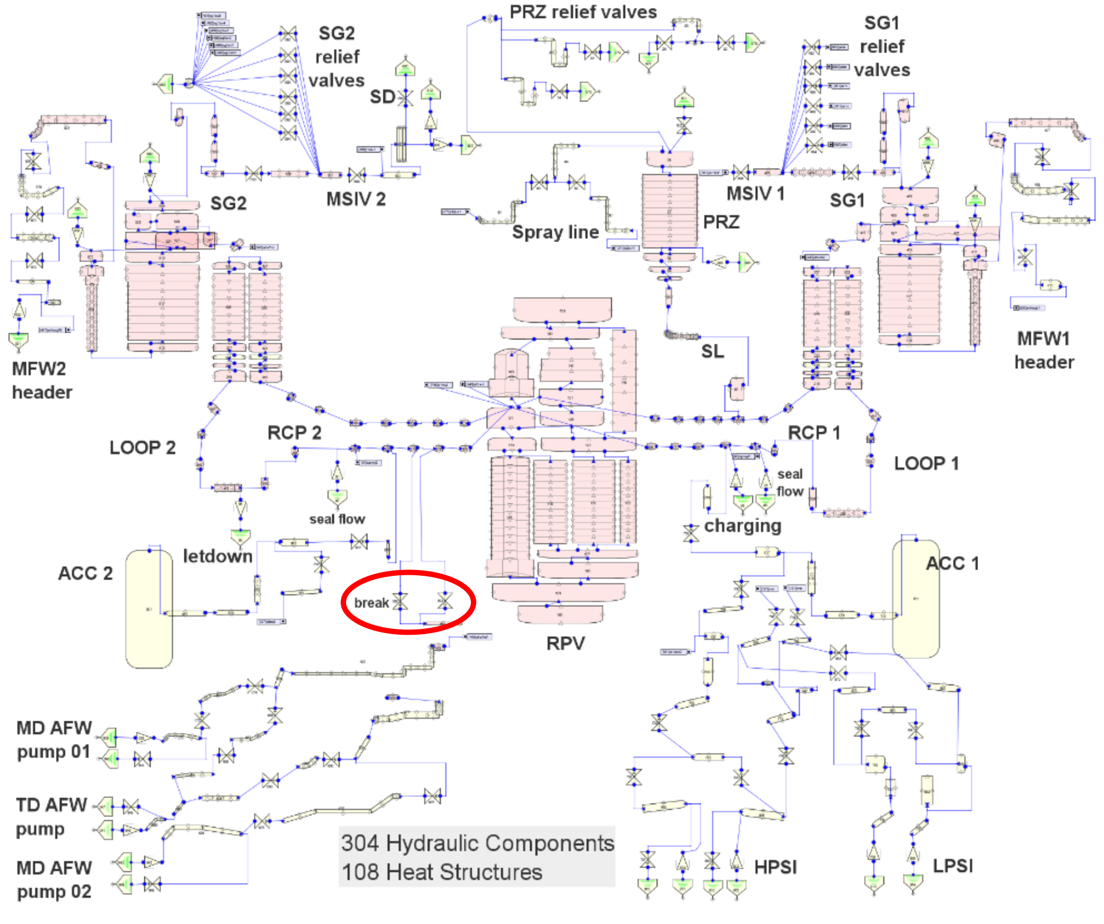


Figure 1. PWR nodalization scheme – SNAP hydraulics component view

The primary side includes the RPV and the two primary loops (loops 1 and 2), both with a reactor coolant pump (RCP) and steam generator (SG) U-tubes. The pressurizer (PRZ) vessel is connected to its spray lines – two power operated relief valves (PORVs) and two safety valves – and to the primary loop 1 through the surge line (SL). The two trains of emergency core cooling system (ECCS) comprise active high (HPSI) and low (LPSI) pressure safety injection pumps and accumulators (ACCs). The secondary side consists of the steam generators with main (MFW) and auxiliary (AFW) feedwater systems, and main steamlines with SG relief valves and main steam isolation valves. Finally, the break is modelled with two valves connected to the cold leg (and volume after the valve to collect discharged mass), which allows to model single and double-ended breaks. In our case one valve opens at accident initiation. Additional details of the RELAP5 model can be found in Ref. [7] and references therein.

A set of 4 LOCA scenarios due to breaks at the cold leg of sizes equal to 2.54 cm (1 inch), 7.62 cm (3 inch), 15.24 cm (6 inch) and 30.48 cm (12 inch) are considered in this work. For convenience, break sizes are designated with units of inch throughout the text. The breaks are assumed approximately at the last weld of the cold leg with the RPV (Fig. 1). The breaks occur at 16.7 min under steady-steate, normal operating conditions (pressure of 15.5 MPa and hot/cold leg temperatures of 324/286°C). Loss of offsite power is assumed with only 1 ECCS train available. ECCS water temperature of 27°C is considered. We also assume hot full power conditions at 100% with ANS79 decay heat model. The reactor trips on low PRZ pressure signal.

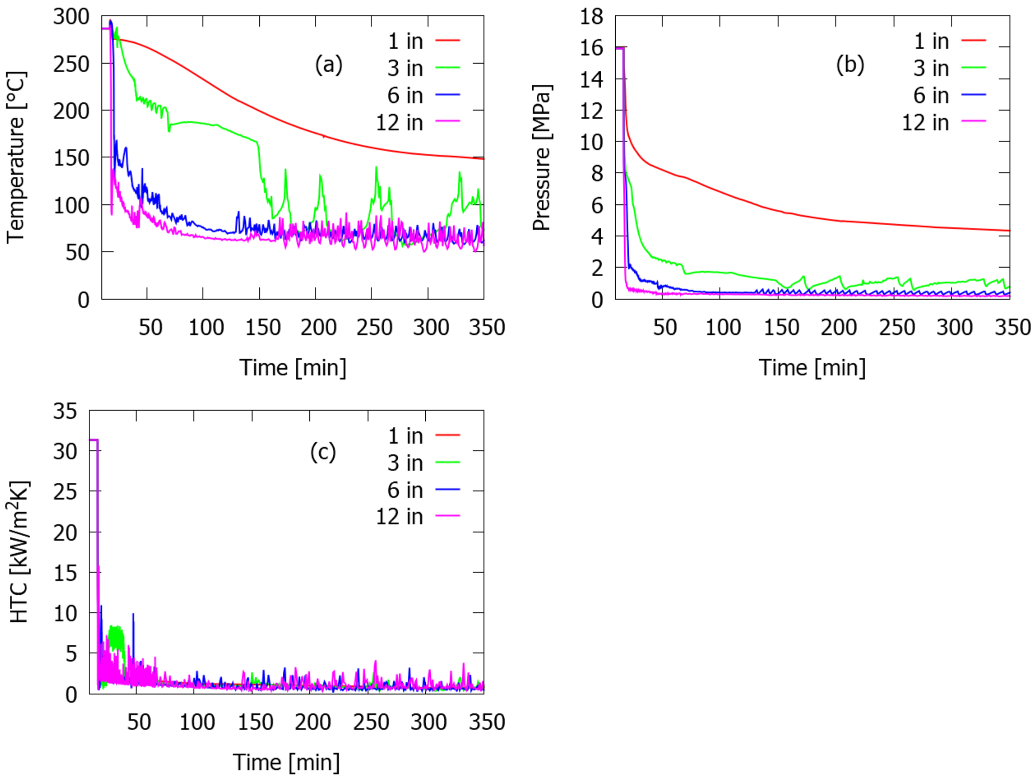


Figure 2: For the 4 cold leg break sizes considered, time histories of (a) fluid temperature, (b) fluid pressure and (c) heat transfer coefficient between fluid and RPV wall

The TH results relevant for the subsequent analyses are presented in Fig. 2. These include the time histories of fluid temperature and pressure, and heat transfer coefficient (HTC) between the fluid and RPV wall, at the downcomer cell located approximately in the middle of the core height. This is the location with the highest axial neutron fluence and it is expected to result in the highest embrittlement of the RPV beltline material. As one could expect, Fig. 2 shows that temperature and pressure drop rates increase with break size. Rather long period fluctuations in temperature can be observed for the 3 in break. These could be ascribed to interaction of core heat transfer phenomena, changes in primary pressure (also observed in Fig. 2-b) and intermittent injection of LPSI pumps.

3 STRUCTURAL INTEGRITY ANALYSES

The TH results are employed as inputs in the subsequent thermo-mechanical and fracture mechanics analyses to obtain the temperatures, stresses and mode I stress intensity factors (SIF or K_I) of postulated cracks in the RPV wall. The analyses are performed with the Fracture Analysis of Vessels – Oak Ridge (FAVOR) code [8], which has been specifically developed to perform integrity analyses of RPV under PTS. FAVOR employs an axisymmetric 1D simplification of the RPV wall in the thickness (radial) direction so, only 1 point input data for each transient is required. Liner-elastic RPV material response is assumed in FAVOR.

3.1 Thermo-mechanical analyses

The RPV considered in this work has an inner radius of r_i = 1676 mm and wall thickness of w = 173.8 mm, of which 5.6 mm is austenitic stainless-steel cladding and the rest is low-alloy ferritic steel (base material). The temperature-dependent material properties of both steels from Ref. [4] are considered in this work. The fluid temperatures and HTC obtained with RELAP5 are employed in the heat transfer analyses to obtain the RPV wall temperatures. Together with the fluid pressure, these are used in the mechanical analyses to compute stresses.

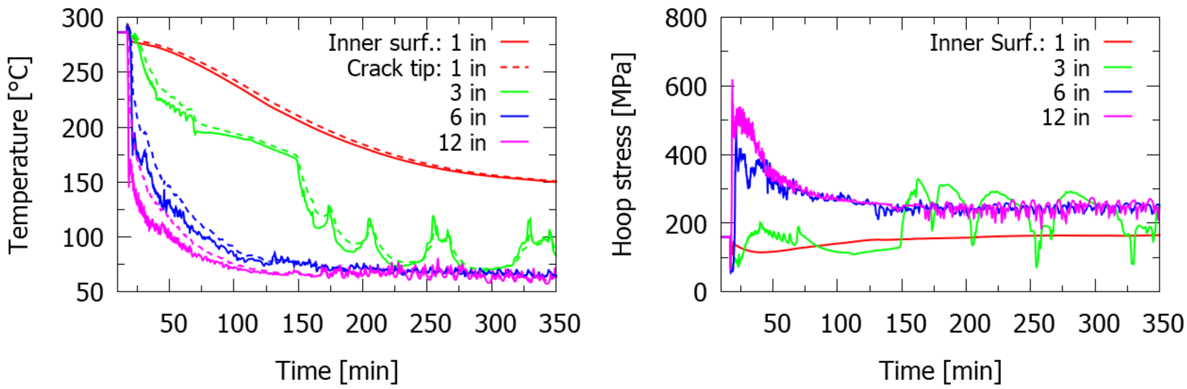


Figure 3: For the 4 cold leg break sizes considered, (left) inner surface and crack tip (a/w = 0.1) temperature histories and (right) hoop stress histories of the RPV inner surface

Figure 3 shows the resulting time-histories of temperatures and hoop stresses at inner surface of the RPV. Comparing both results one can observe that higher temperature-drop rates produce higher stresses. The fluid temperature oscillations noticed in Fig. 2 are also observed at the RPV surface. Increases of temperature result in drops of stress levels and vice versa.

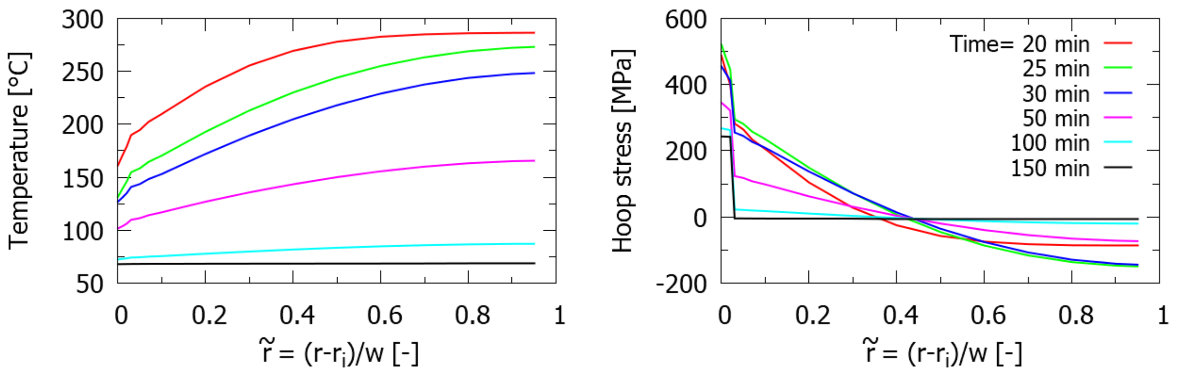
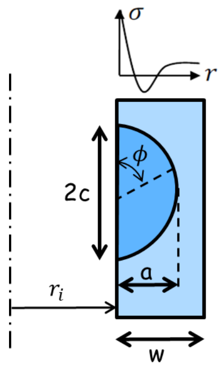


Figure 4: Timely profiles along the RPV wall thickness of (left) temperatures and (right) hoop stresses for the break size of 12 inch

Figure 4 presents the timely profiles through-thickness of the RPV wall temperatures and hoop stresses obtained for the 12 in break transient. A sudden change in the profiles' slope can be noticed due to the different cladding and base material properties. As observed also in Fig. 3, the stress levels are rather high due to the linear-elastic material response. This conservative assumption becomes realistic for aged RPVs, as the yield strength increases with irradiation.

3.2 Fracture mechanics analyses

In the fracture mechanics analyses, the SIFs (or crack driving forces) at the crack tip are computed for different semi-elliptical cracks postulated at the inner surface of the RPV (Fig. 5). FAVOR computes the crack SIFs by a weight function method using the stress profiles obtained in the thermo-mechanical analyses.



- $K_I \;= \sum_{i=0}^{3} C_i \sqrt{\pi a}\, G_i$: weight function calculation of SIF

where C_i coefficients are obtained from:

- $\sigma(\tilde{r}) = \sum_{i=0}^{3} C_i \left(\frac{\tilde{r}}{a}\right)^i$: 3rd order polynomial fit of stress profile

for $0 \le \; \tilde{r} \, = \, (r - r_i)/w \, \le \, a$ and where

- $G_i = f(r_i/w\,, a/w\,, 2c/a\,, \phi)$: influence functions obtained from finite element simulations

Figure 5: Sketch of an axially oriented, semi-elliptical, inner surface crack in a cylinder and SIF calculation method by FAVOR

Figure 6-left presents the SIF histories obtained for the 4 break sizes assuming an axially oriented, semi-elliptical crack at the inner surface with depth-to-wall thickness ratio of a/w = 0.1 and length-to-depth ratio of $2c/a$ = 6. One may observe that the SIF for the smallest break size of 1 in is most of the time higher than for the other breaks after ~75 min into the transient due to the available fluid pressure (Fig. 2).

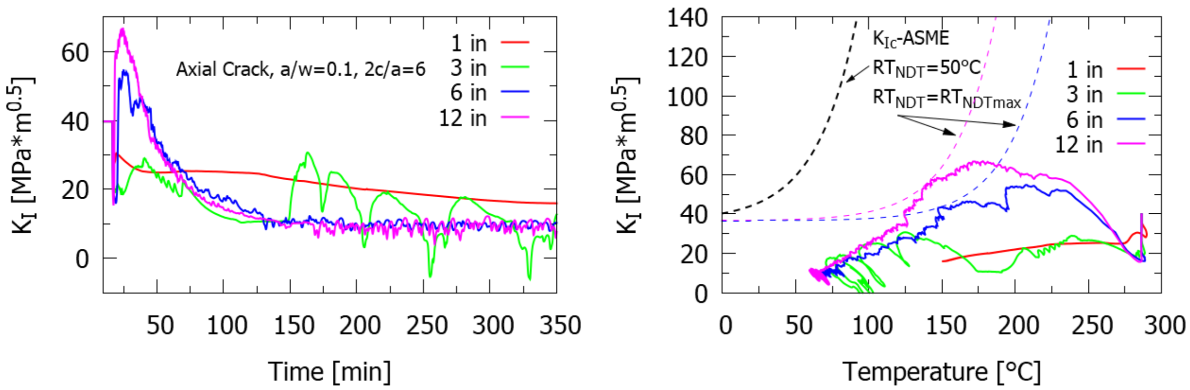


Figure 6: For the 4 cold leg break sizes considered and assuming an axially oriented, semi-elliptical, inner surface crack of depth equal to 10% of wall thickness and length-to-depth ratio equal to 6, (left) SIF histories and (right) SIF vs crack tip temperature curves (solid lines) compared to ASME fracture toughness curves for different RT_{NDT} (dashed lines)

The integrity analysis of the cracked RPV under a PTS event requires to compare the local crack tip conditions (SIF and temperature), also known as the crack loading path, with the fracture toughness curve (or resistance to crack propagation) of the material. For the latter, we use the fracture toughness curve of the American Society of Mechanical Engineers (ASME):

$$K_{IC}(T) = 36.52 + 22.81\exp[0.036(T - RT_{NDT})] \tag{1}$$

where T [°C] is the temperature and RT_{NDT} [°C] is the index reference (Nil-ductility or ductile-to-brittle transition) temperature, which is representative of the embrittlement level of the material, i.e., higher the RT_{NDT}, higher the embrittlement. The units of $K_{IC}(T)$ are [MPa*m$^{0.5}$].

In Fig. 6-right, the local crack-tip conditions are represented by K_I versus crack tip temperature curves (solid colored lines) for the 4 breaks considered. Note that K_I is that from Fig. 6-left and the crack tip temperature is also given in Fig. 3-left (dashed lines). Therefore, the same axial, semi-elliptical crack with a/w = 0.1 and $2c/a$ = 6 is considered here. These curves are compared with the ASME fracture toughness curve, Eq. (1), assuming an arbitrary value of RT_{NDT} = 50°C (dashed black line in Fig. 6-right). If $K_I(T)$ > $K_{IC}(T)$, the postulated crack will initiate its growth in brittle fracture. This is clearly not the case for the studied transients, the assumed value of RT_{NDT} and crack size.

3.3 Temperature margin evaluation

The PTS Rule, as set forth in Title 10, Section 50.61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events," of the Code of Federal Regulations (10 CFR 50.61) and the regulatory guide 1.99 of the United States Regulatory commission (USNRC, RG-1.99) [9] provide the instructions on how to estimate the value of RT_{NDT} as:

$$RT_{NDT} = RT_{NDT(U)} + \Delta RT_{NDT} + M \tag{2}$$

where $RT_{NDT(U)}$ is the reference temperature in the unirradiated condition, ΔRT_{NDT} is the transition temperature shift due to irradiation and M is a margin to account for uncertainties in previous two values. ΔRT_{NDT} can be obtained from surveillance data or estimated as a function of fluence and alloying (copper, nickel) content of the RPV (weld and base) materials.

The distance (in temperature units) between the toughness curve and the crack loading path in Fig. 6-right for a given RT_{NDT} is the so-called PTS temperature margin. It is clear from Eq. (2) that, given $RT_{NDT(U)}$ and M, the temperature margin assessment is in fact a matter of finding the embrittlement margin (in terms of temperature shift, ΔRT_{NDT}) for a given transient and crack. Because ΔRT_{NDT} correlates with neutron fluence, so it does with operational time. Evaluation of temperature margin requires finding the minimum temperature where both curves (toughness and loading path) meet, i.e., $K_I(T)$ = $K_{IC}(T)$, named as the classic "tangent point" approach. This is represented in Fig. 6-right by the coloured dashed lines. The condition $K_I(T)$ = $K_{IC}(T)$ for margin calculation and, thus, $K_I(T)$ > $K_{IC}(T)$ for crack growth initiation, neglects the warm pre-stress (WPS) effect. WPS effect stands for the experimental observation that stressing a crack at high temperatures during the beginning of the transient, while the material is ductile, results in an apparent increase of K_{IC} at lower temperatures later in the transient, when the material may become brittle. Thus, no crack growth initiation will occur if K_I exceeds K_{IC} by some amount when WPS effect is considered [5]. However, consideration of WPS effect is still controversial and not generally accepted. Therefore, WPS effect is not considered in this paper and the "tangent point" approach is used.

Examples of $RT_{NDT(U)}$ and M values are -20°C [10] and -5°C [9], respectively. However, these values may vary depending on the material, production and position in component. In this paper, the evaluation of temperature margin is performed in a more general way by finding the

RT_{NDT} maxima ($RT_{NDT_{max}}$), i.e. inclusive of $RT_{NDT(U)}$ and M, for which the "tangent point" approach conditions are met for the 4 transients studied and for different crack sizes.

In Figure 7, the results of $RT_{NDT_{max}}$ as a function a/w for $2c/a$ ratios of axial, semi-elliptical, inner surface cracks are presented for the transients of 6 inch and 12 inch break sizes. Note that $RT_{NDT_{max}}$ results for the 1 inch and 3 inch break transients are either the initial RPV temperature or infinite, due to $K_I(T)$ being, respectively, equal or below 36.52 MPa*m$^{0.5}$ (minimum value of K_{IC}, Eq. (1)), and meaning that these transients are not a PTS concern.

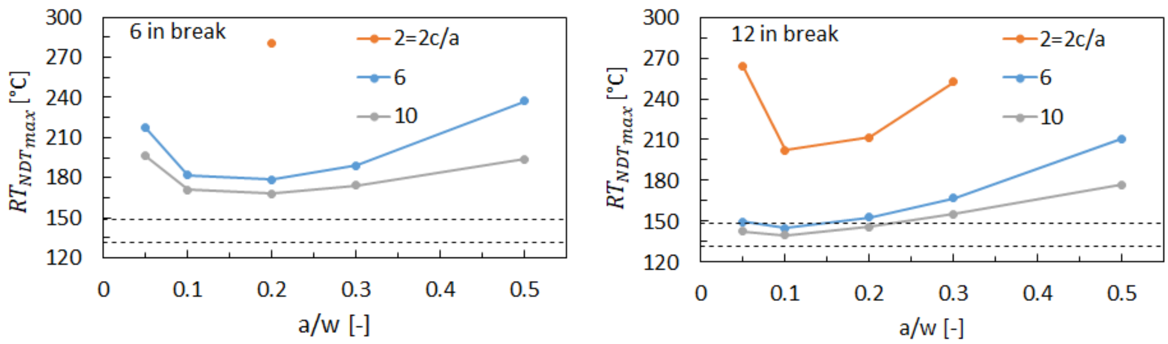


Figure 7: RT_{NDT} maxima (solid lines) as a function of crack depth and different crack length-to-depth ratios obtained for the (left) 6 inch and (right) 12 inch breaks assuming axial, semi-elliptical, inner surface cracks. Dashed lines indicate regulatory screening limits

The results in Fig. 7 indicate that $RT_{NDT_{max}}$ decreases with increasing break size and crack length. The minimum values of $RT_{NDT_{max}}$ are found for a/w = 0.2 and 0.1 for the break size transients of 6 inch and 12 inch, respectively. Conservative estimations of RT_{NDT} after 60 years of full power operation of new reactors are about 60°C [10]. One could possibly argue that the $RT_{NDT_{max}}$ results in Fig. 7, all above ~139°C, indicate that there is sufficient margin for the safe and long-term operation of the plant under the studied transients. On the other hand, the screening limits provided in 10 CFR 50.61 restrict the maximum values of RT_{NDT} during the operational life to 132°C for axial welds, plates, and forgings, and 149°C for circumferential welds. The results in Fig. 7-right thus show that cracks with a/w = [0.05, 0.15] and $2c/a$ = [6, 10] subjected to the 12 inch break transient have lower $RT_{NDT_{max}}$ than the regulatory limits (included in Fig. 7 as dashed lines) for cracks placed in circumferential welds. Therefore, this could imply that an operating RPV with RT_{NDT} near the screening limit having such cracks and subjected to this transient could undergo brittle propagation. However, the rather small difference of about 10°C between $RT_{NDT_{max}}$ and the regulatory limit could be overcome by reducing analysis conservatism through consideration of WPS effect in margin evaluations. Additional analyses following-up this work could include transients originating from larger break sizes and at the hot leg.

4 CONCLUSIONS

In this paper, the preliminary PTS analyses of a PWR under LOCA transients are presented together with the methodology to perform temperature margin evaluations. The analyses assume cold leg breaks of 4 different sizes as the initiating events leading to PTS. The TH analyses are performed with RELAP5 and provide the inputs to the structural analyses performed with FAVOR. The results indicate that, for the transients considered, larger break sizes produce higher temperature-drop rates, higher stress levels and, typically also, higher SIFs of postulated cracks at the RPV inner surface. The temperature margin evaluation is performed following the "tangent point" approach in terms of the maximum RT_{NDT} to avoid brittle fracture propagation. The results show that lower $RT_{NDT_{max}}$ are obtained for increasing break size transients and for longer cracks. Cracks depths equal to a/w = 0.2 and 0.1 yield $RT_{NDT_{max}}$ minima, respectively, for 6 inch and 12 inch breaks. While the $RT_{NDT_{max}}$ results suggest that these are sufficient for the safe operation of the plant, ranges of axial crack depths and lengths located in circumferential welds of the RPV have been found for which the $RT_{NDT_{max}}$ values are up to 10°C lower than regulatory screening values. Future work will be directed towards consideration of additional break sizes and locations in the primary system and to include WPS effect in temperature margin evaluations.

ACKNOWLEDGMENTS

The authors gratefully acknowledge the financial contribution of the Slovenian Research Agency through the research program P2-0026.

REFERENCES

[1] IAEA, "Pressurized Thermal Shock in Nuclear Power Plants: Good Practices for Assessment", IAEA-TECDOC-1627, INTERNATIONAL ATOMIC ENERGY AGENCY, 2010.

[2] M. Kirk, "Development of the Alternate Pressurized Thermal Shock Rule (10 CFR 50.61a) in the United States", Nuclear Engineering and Technology, vol. 45, 2013, pp. 277-294.

[3] M. Niffenegger, et al., "Uncertainties in Pressurized Thermal Shock Analyses", ASME Pressure Vessels & Piping Conference (PVP2019). PVP2019-94076, San Antonio, TX, USA, 2019.

[4] D. F. Mora, et al., "Fracture mechanics analyses of a reactor pressure vessel under non-uniform cooling with a combined TRACE-XFEM approach", Engineering Fracture Mechanics, vol. 238, 2020, pp. 107258.

[5] M. Zarazovskii, et al., "Impact of the outer surface air cooling and WPS approaches on the brittle fracture margin of WWER RPV", ASME Pressure Vessels & Piping Conference (PVP2020). PVP2020-21736, Minneapolis, Minnesota, USA, 2020.

[6] USNRC, RELAP5/MOD3.3 Code Manual, vol. 1-8, patch 05, , Information Systems Laboratories, Inc., Rockville, MD, USA, 2016.

[7] A. Prošek, M. Matkovič, "RELAP5/MOD3.3 Analysis of the Loss of External Power Event with Safety Injection Actuation", Science and Technology of Nuclear Installations, vol. 2018, 2018, pp. 6964946.

[8] P. T. William, et al., Fracture Analysis of Vessels – Oak Ridge FAVOR, v16.1, Computer Code: Theory and Implementation of Algorithms, Methods, and Correlations, ORNL/LTR-2016/309, 2016.

[9] USNRC, "Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials", 1988.

[10] USNRC, "AREVA Design Control Document Rev. 5 - Tier 2 Chapter 05 - Reactor Coolant System and Connected Systems - Section 5.3 Reactor Vessel", https://www.nrc.gov/reactors/new-reactors/design-cert/epr/reports.html, 2013.